VERIFICATION and VALIDATION of CORE ANALYSIS CODE SYSTEM SAPFIR_VVR95&RC

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ABSTRACT

Core analysis code system SAPFIR_VVR95&RC intended for the thermal water-moderated water-cooled reactors (VVR) neutronic calculation in process of burn-up. V&V approach and results are presented. The code system has GAN certificate for propulsion reactors.

I. INTRODUCTION

The core analysis code system contains two key units - cell calculation code SAPFIR_VVR95 with the library BNAB-78/C-95 and the universal three-dimensional diffusion mesh code RC.

The main unit of this code system is SAPFIR_VVR95. The last is a code for the neutron-and physical calculation of cells in thermal water moderated water-cooled reactors in the process of burn up formed on base of SAPFIR package. The package SAPFIR[1] was designed as basis for formation of higher accuracy codes intended for checking and research calculations. The versatility of this package results from the following:

– only evaluated nuclear data files are utilized as initial nuclear- physical information, and files-to-working libraries conversion codes are included in the package architecture;
– there are no fundamental restrictions on the composition and geometry of the systems under consideration which are present in nuclear reactors;
– there are no models with adjustable parameters as important components.

This is due to the use of algorithms:

– adequate representation of basic nuclear data in the package[2] working libraries;
– generalized subgroup approach (GSA) in resonance energy region[3];
– termalization in 40-microgroups [4];
– burn-up of nearly 70 isotopes [5];
– calculation of first collisions probabilities (FCP) by the method[6] based on the geometrical module SCG-5 [7] (with FCP approximation[8]).

In addition, the package SAPFIR includes a set of segments for benchmark calculations using both Russian (Databank of «Kurchatov Institute»[9]) and foreign (ENDF/B type) nuclear data libraries. This makes it possible to prepare a set of test tasks during verification. These tasks are based on

– the solution of the neutron slowing down equation with detailed representation of cross-sections in the resolved resonances region;
– the solution of the neutron transport equation in cells and reactors by Monte Carlo method and method of discrete ordinates;
– the solution of the neutrons diffusion problem in many groups with finer finite difference mesh.

Verification and validation of such code system discussed in the next sections.

II. VERIFICATION AND VALIDATION (V&V) APPROACH

The codes system has been verified with regard to recommendations by GAN. The set of the tests was selected in view of the basic verification activities such as[10]:

– checking the constants validity;
– checking that mathematical models and calculation algorithms are adequate to physical problem;
– definition of application area for the models and algorithms implemented;
– assessment of the accuracy of calculated neutron-physical characteristics important to reactor safety in the application area of the code system.

For practical application, of the first importance is the last task in the list, but to accomplish it, the first three problem must be solved, because the final error of computed integrated characteristics of the reactor includes a number of components such as:

– uncertainty of the nuclear data utilised;
– inaccuracy of models for description of physical processes used in the execution of the spectral task (task of neutrons transport with regard to moderation, resonance absorption, thermalization in a real reactor cells geometry);
– inaccuracy of homogenisation and neutrons diffusion models for reactor;
– inaccuracy of models for description of the feedback related to temperature and change in isotope composition due to reactor burnup and poisoning;
– uncertainty in initial data due to technological assumption.

As a result, V&V problem of SAPFIR_VVR95&RC gave us an approach based on the use of system of the tests implemented in sequence:

– Numerical and experimental tests of a "Nuclear Data Benchmark" type intended for a justification of the nuclear data and models utilised in spectral tasks;
– Numerical and experimental tests of "Separate Effects Benchmark" type intended for justification of specified neutron-physical models taking into account change in isotope composition, fuel and coolant temperature, absorbers effects etc.;
– Numerical and experimental tests on facility with fuel elements and assemblies, similar in properties to the certain reactor types ("Core Physics Benchmark");
– Results of measurements of neutron-physical characteristics in full-scale cores at the beginning of campaign (without power);
– Measured neutron-physical characteristics in full-scale cores of the operating reactor for overall codes system V&V with the account for modelling operation at power, poison, burn-up.

Some V&V results for some type tests are given below.

III. V&V RESULTS

Below some items of V&V matrixes for reactor types in question are presented:

– V&V of models for calculation a neutron spectrum and resonance absorption, first of all for ^{238}U with a variation of enrichment, a lattice pitch, fuel temperature (Doppler effect);
– V&V of models for fuel burn-up and Pu build-up in cell calculation;
– V&V of algorithms for cell and reactor calculation codes simulating refuelling with reactivity compensation by a liquid absorber.
– V&V of burn-up models for «heavy» absorbers;
– V&V of algorithms for solving transport equation for complicated geometry cells with non-uniformity effects such as heavy absorbers, different types of fuel rods;
– V&V of models and algorithms for calculation of neutron and power density in core non-uniform in radius and height;
– assessment of validity of calculated reactivity effects;
– V&V of joint neutron-physical and thermal-hydraulic models.

A. SAPFIR_VVR95 V&V on tests and experiments "Nuclear Data Benchmark" type

The main objective of these tests group was to check nuclear data libraries. Both the calculation results of mathematical tests and the results of comparison with experimental data (modeling in buckling approximation) were analyzed. Some comparison results are given in Table 1.

Table 1 Spectral indexes in the numerical tests

Test	Code	k_∞	ρ^{28}	δ^{25}	δ^{28}	CR
NB-1	Monte Carlo[1]	1.1449	1.363	0.0803	0.0722	0.798
		(.14)	(.6)	(1.1)	(.6)	(.4)
	SAPFIR	1.1461	1.3573	0.0795	0.0689	0.7949
NB-2	Monte Carlo [2]	1.1748	2.612	1.51-1	2.97-1	2.148
	SAPFIR	1.1755	2.559	1.48-1	2.81-1	2.115
NB-4	Monte Carlo	1.3424	2.654	0.159	0.0617	0.549
		(.26)	(.6)	(.6)	(.8)	(.3)
	SAPFIR	1.3433	2.647	0.1547	0.0599	0.5496
NB-5	Monte Carlo	1.1456	8.503	0.5480	0.133	1.006
		(.15)	(.8)	(1.7)	(.5)	(.3)
	SAPFIR	1.1395	8.490	0.5443	0.1295	1.0065

B. Моделирование экспериментов типа "Separate effect benchmark"

For these tests group, we considered results of modeling for experiments and numerical tests, as well as calculation accuracy for separate physical effects such as:

- temperature Doppler effect;
- temperature effect (full);
- void effect due to change in water density down to complete dryout;
- fuel burn-up effects (Yankee Reactor, VVER-440, BWR lattice, mixed fuel).

As an example of test tasks calculations in this tests group we present calculation of Doppler effect, void and burn-up effects.

1. Doppler effect modelling in SAPFIR_VVR95

This section presents results of Doppler effect calcu-

[1] (SD) - standard deviation in %

[2] The uncertainty in the Monte Carlo values is not available

lations in the mathematical test described in detail in [11,12]. Table 2 shows the results of k_∞ calculation at different temperatures. Table 3 presents the reactivity variation for the given temperature range.

Table 2 Multiplication factor vs temperature

Enrichment (w.%)	Temperature Water	Fuel	MCNP	SAPFIR VVR95	Deviation in reactivity
0.711	300	300	---	0.6077	
1.6	300	300	0.9177(33)	0.9184	0.08%
2.4	300	300	1.0756(37)	1.0733	-0.20%
3.1	300	300	1.1597(42)	1.1636	0.29%
3.9	300	300	---	1.2382	
0.711	600	600	0.6638(6)	0.6653	0.34%
1.6	600	600	0.9581(6)	0.9607	0.28%
2.4	600	600	1.0961(7)	1.0987	0.22%
3.1	600	600	1.1747(6)	1.1768	0.15%
3.9	600	600	1.2379(6)	1.2400	0.14%
0.711	600	900	0.6567	0.6575	0.19%
1.6	600	900	0.9484	0.9507	0.26%
2.4	600	900	1.0864	1.0881	0.14%
3.1	600	900	1.1641	1.1658	0.13%
3.9	600	900	1.2271	1.2284	0.09%

Table 3 Doppler effect

Enrichment (wt%)	Temperature range	MCNP	SAPFIR
0.711	(600-900)°K	-0.0163(23)	-0.0178
1.6	(600-900)°K	-0.0108(9)	-0.0110
2.4	(600-900)°K	-0.0081(8)	-0.0089
3.1	(600-900)°K	-0.0078(7)	-0.0080
3.9	(600-900)°K	-0.0071(6)	-0.0076

2. The NEACRP HCLWR burn-up benchmark problem

The test task for calculation of burn-up with uranium-plutonium fuel [13] in closely lattices is the rigid test for the spectral codes. The test is of special interest because available are numerous calculated results of nearly all known neutron-physical codes. The SAPFIR_VVR95 calculations are carried out for different water density (void fraction) in cells at the onset of the campaign. For initial void fraction burn-up is calculated to 50 GWt*day/t.

As an example Table 4 compares concentration of key isotopes obtained from burn-up calculation in SAPFIR_VVR95 and DANGE code (70 groups, JEF-1 library, block factors in resonance region). For DANGE burnup results are present in more detail.

3. Power reactor depletion experiments

Experimental data from Yankee-reactor, VVER-365, VVER-440 were used for V&V of burn-up models in SAPFIR_VVR95&RC. Calculated results of burnup-dependent Pu isotopes (^{239}Pu/^{240}Pu, ^{240}Pu/^{241}Pu, ^{241}Pu /^{242}Pu) concentration ratios are shown on Fig. 1 in comparison to experimental data [14] for VVER-440.

Table 4 Isotopes concentration

Cell 1 (V_m/V_f=0.6)						
Burnup	30 GWt*day/t		ε%	50 GWt*day/t		
Isotope	SAPFIR	DANGE		SAPFIR	DANGE	ε%
^{235}U	4.096E-05	4.078E-05	0.44	3.086E-05	3.086E-05	0.00
238U	1.959E-02	1.961E-02	-0.12	1.914E-02	1.917E-02	-0.19
^{236}U	5.432E-06	5.602E-06	-3.04	7.435E-06	7.642E-06	-2.71
^{239}Pu	1.446E-03	1.437E-03	0.63	1.387E-03	1.376E-03	0.78
^{240}Pu	6.812E-04	6.877E-04	-0.94	6.740E-04	6.874E-04	-1.95
^{241}Pu	2.798E-04	2.823E-04	-0.89	2.792E-04	2.803E-04	-0.40
^{242}Pu	2.017E-04	1.974E-04	2.16	1.996E-04	1.896E-04	5.28
Cell 2 (V_m/V_f=1.1)						
^{235}U	4.236E-05	4.240E-05	-0.09	3.234E-05	3.239E-05	-0.15
238U	1.996E-02	2.005E-02	-0.44	1.958E-02	1.968E-02	-0.48
^{236}U	5.130E-06	5.233E-06	-1.98	7.127E-06	7.261E-06	-1.84
^{239}Pu	1.124E-03	1.124E-03	0.02	1.003E-03	1.007E-03	-0.38
^{240}Pu	5.607E-04	5.675E-04	-1.20	5.247E-04	5.389E-04	-2.65
^{241}Pu	2.921E-04	2.897E-04	0.81	3.022E-04	2.976E-04	1.53
^{242}Pu	1.811E-04	1.750E-04	3.44	1.858E-04	1.726E-04	7.68

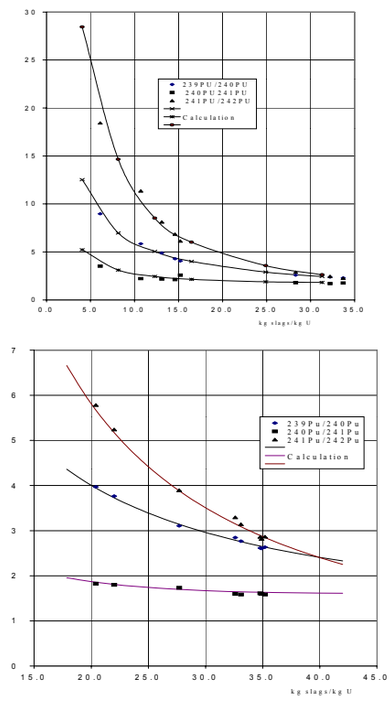

Fig. 1 Pu isotopes concentration ratios vs burnup depth

4. Temperature reactivity factors in facility ZR-6 experiments.

Fig. 2 and Fig. 3 show the results of modeling of experiments[15] for temperature reactivity factor definition on facility ZR-6. Calculations are carried out in 3D geometry.

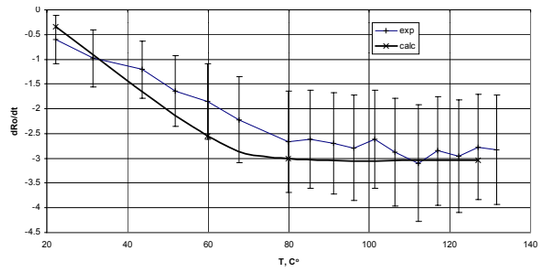

Fig. 2 Temperature reactivity factor (C_B=0.0)

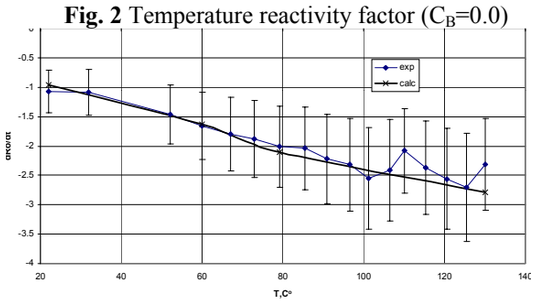

Fig. 3 Temperature reactivity factor (C_B=5.8 g/kg)

C. Numerical and experimental tests «Core Physic Benchmark» type

Tests of this type characterise the code application area for different reactor types, gives characteristics of different fuel assembles, as well as results of comparison to benchmark calculations obtained from SAPFIR_VVR95&RC verification in this type tests. As an exampleTable 5 illustrates results of ZR-6 modeling [15].

Table 5 Facility ZR-6 modeling results.

Assembly		Num. of pin	Water level	k_{eff}
1.27/1.6/ 0.0*)	160/160	625	85.8	0.9978
1.27 /3.6/ 0.0	154/154	805	62.3	1.0021
1.27 /4.4/ 0.0	164/110	625	83.8	1.0020
1.1 /3.6/ 0.0	39/39	1675	98.1	0.9980
1.5/3.6/ 0.0	212/212	409	98.95	1.0012
1.905 /3.6/ 0.0	207/207	349	70.67	1.0033
1.905 /3.6/ 0.0	206/206	409	52.3	1.0032
1.27 /3.6/ 4.0	161/161	1765	48.57	1.0002
1.27 /3.6/ 5.8	163/163	1765	70.12	1.0010
K91 B_4C**)	103/103		80.8	1.0008
K91 B_4C	109/103		98.36	1.0051
K91 Eu	120/116		32.49	1.0010
K91 B_4C	114/114		76.05	1.0025
K271 B_4C	130/26		76.05	0.9992
Average value				1.0012
Standard deviation				0.0020

Also as an example Fig. 4 and Fig. 5 show the results of pin power calculation in comparison with Monte-Carlo methods.

Table 6 gives characteristics of different fuel assembles, as well as results of comparison to benchmark calculations obtained from SAPFIR_VVR95&RC verification in this type tests.

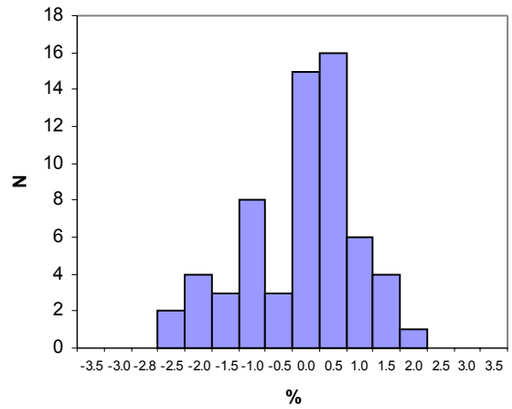

Fig. 4 Deviation in pin power density (VVER-1000)

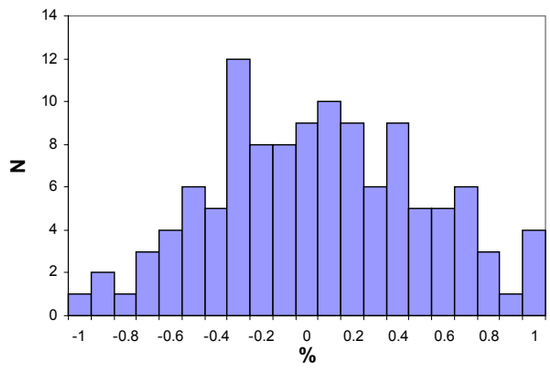

Fig. 5 Deviation in pin power density (VVER-440)

D. V&V on fullscale cores without power and at operation.

Verification experiments were selected such as to cover the range of core basic states, on the one hand, and to achieve these states in different ways, on the other hand. Thus, we can obtain information on the correctness of separate effect modeling (efficiency of control rods, temperature effect, etc.) by comparing calculation results for unloading, control rod (CR) compensation and heat-up condition.

Experimental data used for modeling and comparison purposes was obtained by direct measurements with sufficiently high accuracy (in terms of reactivity):

- determination of criticality ~0.0001 k_{eff};
- CR critical position ~0.0002 k_{eff};
- coolant temperature ~0.0005 k_{eff};
- activation measurements of power density in as-

*) 1.27 –lattice pitch (см), 1.6 – enrichment (%) 0.0 – B10 concentration, g/l

**) Cassette assemblies

semblies 3%;

- activation measurements of fuel elements power density 3%;

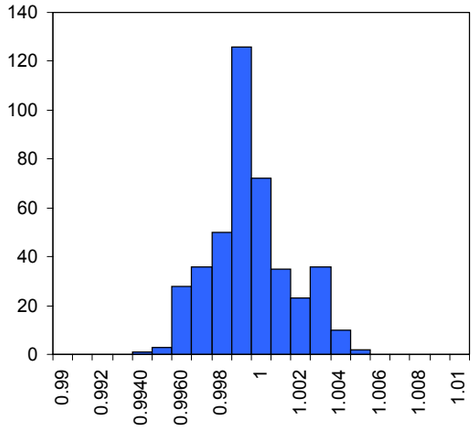

Fig. 6 Deviation of k_{eff}

For modeling power operation, input data of lower though still adequate accuracy is available.

The derived characteristics of the code system accuracy are given on Fig. 6 (in term of k_{eff}) and in Table 7.

These results are derived from more than 500 experiments.

IV. CONCLUSION

From these results, it is concluded that the current code system has sufficient capability for evaluating the core characteristics of any water-moderated water-cooled reactor fuel and core designs.

Core analysis code system SAPFIR_VVR95&RC uses successfully for neutronic and physical calculation water-water propulsion reactors during 10 years. At present time V&V of code system is carried out for reactors of VVER and RBMK type.

Table 6 Water-water reactor features considered in the development and V&V of the SAPFIR_BBR95&RC codes system.

Experiment and facility Designs	Test number (assumed)							
	1[3]	2	3	4	5	6	7	8
1. Fuel:								
1.1. Metal				+		+		
1.2. Ceramic.	+				+	+		
1.3. Disperse.		+	+			+	+	+
2. Assembly:								
2.1.Channel-type		+	+	+	+		+	+
2.2. Hexagon (Cassette)	+							
2.3. Assembly with different fuel elements						+		
3. Absorber elements :								
3.1. Control rods and rings							+	
3.2. Control rods and rings with plates						+		
3.3. Cluster	+							
3.4. Single rod and rings								+
4. Burnable poisons:								
4.1. Rods.			+				+	+
4.2. Bars .			+				+	+
4.3. Assembly elements made of structural materials with poison		+						
4.4. Burnable poison mixed with fuel	+							
5. Absorbers:								
5.1. Pins (rods)						+	+	+
5.2. Ring						+		
6. Heating to the hot state		+			+	+	+	+
7. Change in the coolant density (up to complete dry out) in fuel assembly				+				
Deviations from the benchmark	1	2	3	4	5	6	7	8
1. Multiplication factor								
1.1. Mean square deviation (%)	0.1	0.1	0.1	0.2	0.2	0.2	0.4	0.4
1.2. Maximum deviation (%)	0.2	0.3	0.3	0.6	0.3	0.4	0.7	0.7
2. Power density								
2.1. Mean square deviation (%)	1.7	1.8	2.3	1.4	0.2	1.2	1.4	1.6
2.2. Maximum deviation(%)	4.0	4.0	5.0	3.0	0.3	2.6	2.7	2.8

[3] Tests for VVER-1000 and VVER-440 reactor types

Table 7 Experiments with full-scale cores used in V&V of the SAPFIR_BBP95&RC.

Measured parameters	Without burnup	With burnup
1. Core criticality parameters in the experiments with fuel unloading	1.0003 ±0.0008	
2. Core criticality parameters in the cold state	1.0009 ±0.0008	1.0001 ±0.0008
3. Core criticality parameters in the hot state	1.0001 ±0.0007	0.9999 ±0.0016
4. Core criticality parameters in the experiments with core heat up to operation condition	0.9986 ±0.0007	1.0002 ±0.0015
5. Differential worth of control rods under operation condition	5%	5%
6. Differential worth of control rods during core heat up	7%	7%
7. Temperature reactivity factor	10%	10%
8. Core criticality parameters in experiments with recompensation of control rods	1.0006 ±0.0010	0.9999 ±0.0015
9. Differential worth of control rods at recompensation of control rods	7%	10%
10. Core criticality parameters in experiments with recompensation of some of control rods	1.0017 ±0.0022	
11. Differential worth of some of control rods at recompensation	10%	
12. Worth of control rods in the experiments with drop some of control rods. Core subcriticality	5%	5%
13. Worth of scram rods in the experiments with scram rods drop	10%	
14. Core criticality parameters in the experiments with different reactor power operation modes		0.9988 ±0.0022
15. Core criticality parameters in the experiments with reactor poisoning and overpoisoning with xenon	1.0006 ± 0.0015	0.9990 ± 0.0006
16. Differential worth of control rods in the experiments reactor poisoning and overpoisoning with xenon	7%	
17. Power density peaking factor (k_r, k_z)	3%	
18. Core power density	6%	
19. Axial power density	5%	7%
20. Fuel elements power density	7%	

[1] Artemov V., Elchine A., Ivanov A., Obuchov V., Osipov V., Sergeev V., Tebin V. Software package SAPFIR for reactor cells calculation: *VANT, series Nuclear Reactors Physic and engineering, 4, pp. 68-71.,*1985

[2] Karpov A., Tebin V., Many group neutron data preparation system for SAPFIR package with NJOY-B30 code system. *Proceeding of «Neutronic-95» seminar, Obninsk*., 1995

[3] Tebin V., Judkevich M. Generalized subgroup approach to resonance absorption calculation: *Atomic energy, v.59, is. 2, p. 96,* 1985

[4] Gomin E., Maiorov L. Code TERMAC for group neutron cross-sections in thermalization region *VANT, series Nuclear Reactors Physic and Engineering, 5(27), p. 70,* 1982.

[5] Pisarev P. Modules OTRAWA and SVP for burnup calculations in BMVC and SAPFIR codes *Third conference proceeding «Nuclear technology tomorrow», Leningrad,* 1992

[6] Gomin E., Maiorov L. About FCP calculation in complex geometry systems*, VANT, series Nuclear Reactors Physic and Engineering 8(21), p. 62-69,* 1982

[7] Alexeev N., Gurevich M. Geometry module SCG-5. *Preprint IAE - 5616/4,* 1993

[8] Ivanov A. Annotation of FCP calculation code *VANT, series Nuclear Reactors Physic and Engineering,* 6(43)*, pp. 55-56,* 1984

[9] Judkevich M. Bank of neutron constants of Kurchatov Institute of Atomic Energy: *VANT, series Nuclear data, 5(59), cmp. 3,* 1984

[10] Artemov V., Ivanov A., Piscarev A.,Obuchov V., Shemaev Yu. The set of tests and V&V experience calculation core code system SAPFIR_VVR95&RC on water-water propulsion reactors facilities. *Proceeding of «Neutronic-96» seminar, Obninsk,* 1996

[11] Mosteller R.D. etc. Benchmarking CELL-2 Using a Monte Carlo Method, *NSAC-136*, 1989

[12] Mosteller R.D. etc. Benchmark calculations for the Doppler Coefficient of Reactivity. *N.S.E. 8, p. 437,* 1991

[13] Y.Ishiguro, H.Akie, H.Takano Preliminary Report of HCLWR Cell Burnup Benchmark Calculation, *NEACRP-A-849,* 1988

[14] Stepanov A., Makarova E. et all Burnup and isotope composition determination VVER-440 spent fuel, *Atomic energy, vol. 55, is. 3,* 1983

[15] "Experimental investigation in the physical properties of VVER-type uranium-water lattices", Academia Kiado, Budapest, 1991.